Entergy

Entergy Nuclear Southwest
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751
Tel 504 739 6475
Fax 504 739 6698
aharris@entergy.com

Alan J. Harris
Director, Nuclear Safety Assurance
Waterford 3

W3F1-2001-0034
A4.05
PR

April 12, 2001

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Reporting of Licensee Event Report

Gentlemen:

Attached is Licensee Event Report (LER) 01-003-00 for Waterford Steam Electric Station Unit 3. This report provides details of an Uncomplicated Reactor Trip. This condition is being reported pursuant to 10 CFR50.73(a)(2)(iv)(A).

There are no commitments contained in this submittal. Actions described herein are controlled and tracked via the Waterford 3 Corrective Action Program.

Very truly yours,

A.J. Harris
Director,
Nuclear Safety Assurance

AJH/TNS/ssf
Attachment

cc: E.W. Merschoff, (NRC Region IV), N. Kalyanam, (NRC-NRR), A.L. Garibaldi, lerevents@inpo.org - INPO Records Center, J. Smith, N.S. Reynolds, NRC Resident Inspectors Office, Louisiana DEQ/Surveillance Division

IE22

NRC FORM 366 (1-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Waterford Steam Electric Station, Unit 3	05000-382	1 OF 4

TITLE (4)
. Reactor Protection System Trip caused by Turbine Governor Valve Oscillation

EVENT DATE (5) MO	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REV NO	REPORT DATE (7) MO	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
02	13	01	01	– 003 –	00	04	12	01	N/A	05000
									N/A	05000

OPERATING MODE (9): 1
POWER LEVEL (10): 80

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

20.2201(b)	20.2203(a)(3)(ii)	50.73(a)(2)(ii)(B)	50.73(a)(2)(ix)(A)
20.2201(d)	20.2203(a)(4)	50.73(a)(2)(iii)	50.73(a)(2)(x)
20.2203(a)(1)	50.36(c)(1)(i)(A)	50.73(a)(2)(iv)(A)	[redacted]
20.2203(a)(2)(i)	50.36(c)(1)(ii)(A)	50.73(a)(2)(v)(A)	
20.2203(a)(2)(ii)	50.36(c)(2)	50.73(a)(2)(v)(B)	OTHER
20.2203(a)(2)(iii)	50.46(a)(3)(ii)	50.73(a)(2)(v)(C)	Specify in Abstract below or in
20.2203(a)(2)(iv)	50.73(a)(2)(i)(A)	50.73(a)(2)(v)(D)	NRC Form 366A
20.2203(a)(2)(v)	50.73(a)(2)(i)(B)	50.73(a)(2)(vii)	
20.2203(a)(2)(vi)	50.73(a)(2)(i)(C)	50.73(a)(2)(viii)(A)	
20.2203(a)(3)(i)	50.73(a)(2)(ii)(A)	50.73(a)(2)(viii)(B)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
T N Schreckengast / Shift Manager	(504)-739-6349

C[redacted]

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	[redacted]			COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[redacted] (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On February 13, 2001 with reactor power at approximately 80% power, a secondary transient resulted in a reactor trip. During power escalation following replacement of a Tracker Driver Card on Steam Generator Feed Pump 'A' Speed Controller, turbine valve testing was being performed. Turbine Governor Valve (GV) #3 oscillated twice during testing. The resulting secondary transient caused a reactor trip initiated by a Core Protection Calculator generated Variable Over Power Auxiliary Trip. Emergency Feedwater Actuation Signals (EFAS) 1 and 2 automatically initiated on Steam Generator (SG) Level-Low starting all 3 Emergency Feed Water (EFW) pumps but no flow was delivered to the SGs. This is expected, following a reactor trip. The root cause of this event was determined to be failure of the GV #3 PARC 7300 Card. The circuit card was replaced and sent to the vendor for analysis. This event is reportable pursuant to 50.73(a)(2)(iv)(A) as an automatic actuation of the Reactor Protection System. This event did not compromise the health and safety of the public. This event is not considered a Safety System Functional Failure (SSFF).

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Waterford Steam Electric Station, Unit 3	05000-382	01	- 003 -	00	2 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

REPORTABLE OCCURRENCE

On February 13, 2001 a reactor trip was initiated from approximately 80% power by a Core Protection Calculator generated Variable Over Power Auxiliary Trip due to Turbine Governor Valve #3 oscillating. This event is reportable pursuant to 50.73(a)(2)(iv)(A) as an automatic actuation of the Reactor Protection System. A four (4) hour verbal notification was issued on February 13, 2001 pursuant to 10 CFR50.72(b)(2)(iv)(B).

INITIAL CONDITIONS

At the time this condition was identified, Waterford 3 was operating in Mode 1 at approximately 80% power. No structures or components were out of service that contributed to this event.

EVENT DESCRIPTION

On February 13, 2001, Reactor Power was reduced to approximately 69% to replace a Tracker Driver (NTD) [JA] Card on Steam Generator Feed Pump (SGFP) [JK] 'A' Speed Controller. Power was increased to 80% following completion of repairs to SGFP 'A' Speed Controller. The turbine valve-operating mode was changed from "sequential" to "single" valve-operating mode in preparation for turbine valve testing.

Turbine Governor Valve testing was initiated at 80% power in accordance with OP-903-007, "Turbine Governor Valve Testing". During this test, each governor valve is individually closed. The remaining valves are expected to modulate to maintain turbine load constant.

The Nuclear Plant Operator (NPO) depressed the "Test" pushbutton and verified that the Turbine Control System was in the test mode. Governor Valve (GV) #1 was tested satisfactory with normal response.

GV #2 was selected for testing and closed. EH fluid to this valve was isolated locally to facilitate planned repair of an existing leak. Following repairs, the EH fluid to GV #2 was unisolated and the valve was opened. The valve stroke was satisfactory with normal response.

GV #3 was selected for testing. Upon depressing the "Close" pushbutton, the GV #3 rapidly closed, opened, re-closed, and reopened. The NPO noted the remaining governor valves were attempting to modulate as GV #3 re-closed.

A reactor trip was automatically initiated by a Variable Over Power Trip (VOPT) which is a Core Protection Calculator (CPC) [JC] Auxiliary trip.

The Plant was stabilized in Mode 3 using appropriate Operations Department procedures.

Emergency Feedwater Actuation Signals (EFAS) [JA] 1 and 2 were automatically initiated on Low Steam Generator (SG) levels. This is an expected actuation following a reactor trip due to "shrinkage" of the SG water volume. All 3 EFW pumps automatically started as expected. However, there was no EFW flow delivered to the SGs. EFAS 1 and 2 reset and all EFW pumps were secured.

CAUSAL FACTORS

The root cause of this event was the failure of the GV #3 PARC 7300 Card.

CORRECTIVE ACTIONS

1) Replaced failed GV #3 PARC 7300 Card.

2) Sent failed PARC 7300 Card to vendor for failure analysis.

These corrective actions are controlled and tracked via the Waterford 3 Corrective Action Program.

SAFETY SIGNIFICANCE

The PARC 7300 Card failure caused Turbine Governor Valve #3 to cycle open and close, which caused an increase in power, and reactor trip on Variable Over Power Trip (VOPT). The thermal margin (DNBR) available at the initiation of the event exceeded the amount needed to cover the power increase and reduced thermal margin. The CPC DNBR data over that time frame indicated that the DNBR Specified Acceptable Fuel Design Limit (SAFDL) of 1.26 was never violated; thus, there was no challenge to the integrity of the fuel. This event is bounded by the licensing basis excess load and pre-trip steam line break events. The plant was never outside of the design basis. This event did not compromise the health and safety of the public. This event is not considered a Safety System Functional Failure (SSFF).

SIMILAR EVENTS

No previous reportable occurrences were identified that were attributed to an electronic malfunction in the DEH Controls System.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Waterford Steam Electric Station, Unit 3	05000-382	01	- 003 -	00	4 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

ADDITIONAL INFORMATION

Energy Industry Identification System (EIIS) codes are identified in the text within brackets [ ].